NON-PUBLIC?: N
ACCESSION #: 9306100344
LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK-UNIT 2 PAGE: 1 OF 07

DOCKET NUMBER: 05000446

TITLE: MANUAL REACTOR TRIP FOLLOWING INADVERTENT CLOSURE OF FEEDWATER ISOLATION VALVE CAUSED BY INSTRUMENTATION CHANNEL ERROR
EVENT DATE: 05/04/93 LER #: 93-003-00 REPORT DATE: 06/03/93

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 023

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: D. J. REIMER, MANAGER, SYSTEM TELEPHONE: (817) 897-5584
ENGINEERING

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: RTV MANUFACTURER: E331
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On May 4, 1993, Comanche Peak Steam Electric Station Unit 2 was at approximately 23 percent of rated thermal power after a recent power reduction in preparation for a reactor shutdown from the Remote Shutdown Panel to be performed as part of the initial startup sequence. As a result of a combination of instrument inaccuracies, the feedwater anti-waterhammer permissive was lost, resulting in closure of feedwater isolation valve number 1. With steam generator number 1 at 38 percent level and decreasing, a manual reactor trip was initiated. The cause of the event was instrument error. Corrective action included maintenance on the affected instrumentation channels.

END OF ABSTRACT

I. DESCRIPTION OF REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

An event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System.

B. PLANT OPERATING CONDITIONS BEFORE THE EVENT

On May 4, 1993, just prior to the event, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1, Power Operation, with the reactor operating at approximately 23 percent of rated thermal power. Reactor stable power reduction in preparation for a reactor shutdown from the Remote Shutdown Panel to be performed as part of the initial startup sequence.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

An instrument error existed on the feedwater temperature channel providing input to the feedwater anti-waterhammer permissive logic.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

On May 4, 1993, at approximately 2:45 a.m. CDT, an alarm was received at the main control board indicating that a feedwater anti-waterhammer permissive was not clear. The balance of plant reactor operator (utility,licensed) checked for closure of any feedwater isolation valve (FIV) (EIIS:(ISV)(SJ)), and noted that FIV number 1 was in mid-position.

The unit supervisor (utility, licensed) approached the control board and observed an indicated feedwater flow of 400,000 pounds per hour on the channel providing input to the anti-waterhammer logic. The alternate channel indicated flow of greater than 600,000 pounds per hour. At the same time, a steam flow/feed flow mismatch alarm was received. The unit supervisor directed that the alternate channel be selected, and the mismatch alarm cleared.

Insufficient time was available to complete immediate actions initiated to restore feedwater flow. With steam generator (EIIS: (SG)(SB)) number 1 at 38 percent level and decreasing, a manual reactor trip was initiated at 2:53 a.m. At approximately 4:20 a.m., the Nuclear Regulatory Commission was notified of

the event via the Emergency Notification System as required by 10CFR50.72.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE OR
PROCEDURAL OR PERSONNEL ERROR

Upon receipt of the initial alarm, operating personnel recognized the reduced feedwater flow to steam generator number 1 due to closure of the associated FIV. During event review, it was determined that FIV closure was initiated by a combination of instrument errors which resulted in a loss of the feedwater anti-waterhammer permissive signal. Review of data recorded during the event revealed that closure of FIV number 1 took between three and four minutes. Expected closure time is 5 seconds or less.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

Feedwater temperature instrumentation indicated higher than the actual process temperature due to high resistance in one leg of the resistance temperature detector (RTD) (EIIS:DET)), resulting in partial loss of the feedwater anti-waterhammer permissive signal.

Feedwater flow instrumentation indicated lower than the actual process flow due to an inaccurately sensed pressure drop across the flow venturi meter (EIIS:(FI)). This failure in conjunction with the feedwater temperature error resulted in loss of the feedwater anti-waterhammer permissive signal.

Slow closure of Feedwater Isolation Valve Number 1 due to slow hydraulic control circuit response resulted in continued flow to steam generator number 1 for longer than expected.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

The feedwater temperature instrumentation channel failure was caused by either a poor terminal block (EIIS:(BLK)) connection or high lead resistance at the terminal block.

The feedwater flow instrumentation channel failure was caused by a steam leak at the root valve (EIIS:(RTV)) of the instrument line.

The most probable cause of slow closure of FIV number 1 was partial blockage

of the dump port in the hydraulic control circuit due to the presence of particulate or gelled or crystallized hydraulic fluid.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF
COMPONENTS WITH MULTIPLE FUNCTIONS

Slow response of the FIV hydraulic control circuit inhibits the ability of the FIV to respond to a feedwater isolation signal.

D. FAILED COMPONENT INFORMATION

Edwards Valve, Inc., manufacturer's part number 2751823, Gasket, spiral wound, non-asbestos, Flexite Super, 3/4 inch.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

Closure of FIV number 1: The feedwater waterhammer minimization system is
non-safety grade control circuit designed to reduce the potential for waterhammer damage to the steam generators at low power levels. The circuit is not required to operate to prevent or mitigate any of the events analyzed in Chapter 15 of the CPSES Final Safety Analysis Report (FSAR).

The system utilizes non-safety grade instrumentation channels interfacing with safety grade circuit components to initiate a FIV dose signal on the affected feedwater loop upon loss of the anti-waterhammer permissive. Where the system interfaces with safety class equipment, electrical isolation devices preclude any adverse impact on the safety class equipment from failure of nonsafety equipment.

The circuit responds to flow and temperature instrumentation inputs associated with the feedwater system. All interlocks must be present and a feedwater isolation signal absent to allow the FIVs to open.

Feedwater flow as measured by a flow venturi meter in each feedwater loop must be above the low flow setpoint. Once the flow permissives have been cleared allowing the FIV to open, the FIV can remain open regardless of flow as long as feedwater temperature remains within the setpoint criteria.

Feedwater temperature as measured by RTDs on each main feedwater line must be above the low setpoint, and the difference in temperature measured by

RTDs mounted on the feedwater lines outside containment and RTDs mounted inside containment near the main feedwater nozzles must be within specified limits. Once the temperature permissives have been cleared allowing the FIV to open, the FIV can remain open regardless of feedwater temperature as long as feedwater flow remains above the setpoint.

A combination of non-safety grade instrument errors resulted in a loss of the feedwater anti-waterhammer permissive signal which initiated closure of FIV number 1

Manual reactor trip: Following manual initiation of the reactor trip, the Auxiliary Feedwater System automatically initiated. All related components within the system functioned as required.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Operability of FIV number 1 was verified upon successful completion of response time testing on May 5, 1993, at approximately 9:00 p.m., following troubleshooting activities on the hydraulic control circuit. Duration of component inoperability was approximately 42 hours.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

Loss of anti-waterhammer permissive leading to manual reactor trip: Closure of a FIV leads to a loss of normal feedwater flow to the associated steam generator and a reduction in the capability of the secondary system to remove heat generated in the reactor core. Loss of normal feedwater flow is described in section 15.2.7 of the CPSES FSAR and uses conservative assumptions to demonstrate that the primary system never approaches a departure from nucleate boiling condition.

The FSAR analysis assumes an initial reactor power level of 102 percent and an automatic reactor trip on steam generator low-low level. The loss of feedwater anti-waterhammer permissive occurring on May 4 would not have occurred at a power level significantly greater than 23 percent, and a reactor trip occurring at that power level is bounded by the FSAR accident analysis. It is concluded that the event did not adversely affect the safe operation of CPSES Unit 2 or the health and safety of the public.

Slow closure of FIV number 1: The feedwater system isolation valves and the feedwater bypass system isolation valves are automatically closed to isolate the safety related portion of the feedwater system upon receipt of a steam

generator high-high level signal, a safety injection signal, or reactor coolant system low average temperature coincident with reactor trip. The valves are required to close within five seconds to satisfy the associated Technical Specification surveillance requirement.

Figure omitted.

FSBV = Feed Split Flow Bypass Valve
FRV = Feed Regulating Valve
FRBV = Feed Regulating Bypass Valve
FIV = Feed Isolation Valve
FIBV = Feed Isolation Bypass Valve
FPBV = Feed Preheater Bypass Valve

In addition to the normal control action which will close the main feedwater valves during a reactor trip, a safety injection signal will trip the main feedwater pumps, close the feedwater pump discharge valves, and close the feedwater regulating valves as well as the FIVs. The feedwater regulating valves and FIVs provide redundant isolation capability of the main feedwater lines. It is concluded that slow closure of FIV number 1 would not render inaccurate the conclusions of the CPSES FSAR Chapter 15 analyses, and that the condition did not represent a threat to the ability of the plant to respond to an accident condition.

IV. CAUSE OF THE EVENT

Reactor trip: A reactor trip was manually initiated in response to a decreasing level in steam generator number 1 caused by closure of FIV number 1 upon loss of the feedwater anti-waterhammer permissive signal.

Feedwater isolation: Following the event, a review was performed on data collected by the plant computer. The review revealed that closure of FIV number 1

followed a valid actuation signal caused by a feedwater flow instrumentation error occurring in combination with a pre-existing inaccuracy on the feedwater temperature instrument. The combination resulted in a close signal to the loop 1 FIV.

V. CORRECTIVE ACTIONS

Instrument errors were corrected. The feedwater flow instrumentation channel error was corrected by replacement of the body to bonnet gasket on the leaking instrument line root valve. The feedwater temperature instrumentation channel error was corrected by replacing the RTD and relugging the RTD field termination.

Slow closure of FIV 1 was evaluated. Troubleshooting activities following the event included walkdowns to identify hydraulic fluid leaks or obvious mechanical problems. No problems were identified. Functional capability was verified through stroke testing of all FIVs. Each test was successfully completed with valve closure occurring within 5 seconds.

Continuing evaluation of electrical and hydraulic circuits led to the conclusion that the most probable cause of the slow closure was the result of a blockage in the hydraulic portion of the control circuit. The reservoirs were drained and the oil was examined for the presence of blocking agents such as metallic particulate or gelled or crystallized hydraulic fluid. The filter elements were removed and inspected, and the solenoids were examined for blockage or unusual wear patterns. No problems were identified, and the valve was restored to service.

Additional operating experience with slow closure of FIV 1 during a subsequent reactor trip led to the decision to pursue further evaluation of the problem. The results of that evaluation will be discussed in a future License Event Report.

VI. PREVIOUS SIMILAR EVENTS

There have been no previous reactor trips attributable to the causes identified in this report.

ATTACHMENT 1 TO 9306100344 PAGE 1 OF 1

Log # TXX-93230
File # 10200
Ref. # 50.73
TU ELECTRIC 50.73(a)(2)(iv)

William J. Cahill, Jr. June 3, 1993
Group Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-446
MANUAL OR AUTOMATIC ACTUATION OF ANY
ENGINEERED SAFETY FEATURE
LICENSEE EVENT REPORT 93-003-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 93-003-00 for Comanche Peak Steam Electric Station Unit 2, "Manual Reactor Trip Following Inadvertent Closure of Feedwater Isolation Valve Caused by Instrumentation Channel Error."

Sincerely,

William J. Cahill, Jr.

By:
J. J. Kelley, Jr.
Vice President of Nuclear
Operations

TLH:tg
Enclosure

cc: Mr. J. L. Milhoan, Region IV
Mr. L. A. Yandell, Region IV

Resident Inspectors, CPSES (2)

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